Appendix H
High-Level Waste Forms Comparative Analysis

H.1 METHODOLOGY

This appendix evaluates various plutonium (Pu) forms for potential disposal in a geologic repository. Although a repository site has not yet been recommended for development by the President and approved by Congress, this programmatic environmental impact statement (PEIS) assumes (for analysis purposes only) the existence of a hypothetical repository, managed by the Department of Energy (DOE) Office of Civilian Radioactive Waste Management, at the Yucca Mountain Site in southern Nevada. In accordance with the *Nuclear Waste Policy Act* (NWPA) of 1982, as amended by the NWPA Amendments of 1987 (42 USC 10101), DOE is evaluating the suitability of the Yucca Mountain Site as a potential geologic repository for the disposal of spent nuclear fuel and high-level waste (HLW). Such a repository, if approved under the provisions of the NWPA, would serve primarily as the disposal site for commercial and DOE-owned spent nuclear fuel and HLW. Certain highly radioactive material, which the Nuclear Regulatory Commission (NRC) determines by rule requires permanent isolation, may also be disposed of as HLW in a geologic repository. Such a NRC determination or legislative clarification may be required to dispose of the immobilized forms that would result from the Immobilization Alternatives. Since no waste forms are currently licensed for disposal in an HLW repository, data for forms under consideration in this PEIS for ultimate disposal in a repository are compared to data for those forms currently being evaluated for disposal in an NWPA-licensed repository (that is, commercial and DOE-owned spent nuclear fuel and vitrified HLW). The Environmental Protection Agency has specified that vitrification is the best demonstrated available technology for HLW (55 FR 22627). This approach implies that if the behavior of the Pu forms in a repository is the same or better than the commercial spent nuclear fuel or HLW, and if a repository can be licensed for commercial spent nuclear fuel and HLW, then it is possible that the proposed Pu forms could also be disposed in a repository. [Text deleted.] Due to the great amount of data and information available, U-based commercial spent nuclear fuel and vitrified HLW are used for the basis of the comparison.

If the DOE HLW Program changes its approach for disposal of commercial spent nuclear fuel, if the timeframe for acceptance of forms into the program is significantly delayed beyond Pu disposition requirements, or if the Pu immobilized forms or mixed oxide (MOX)-based spent nuclear fuel resulting from Pu disposition alternatives are determined to be unacceptable to a licensed repository, then DOE would analyze the impacts of continued storage of immobilized Pu or MOX-based spent nuclear fuel in a tiered *National Environmental Policy Act* of 1969 (NEPA) document. Simultaneously, DOE will continue its efforts to site and construct a repository that meets the requirements of the NWPA.

This appendix contains a comparative analysis of five Pu forms: (1) immobilized Pu and other radionuclides in borosilicate glass, (2) immobilized Pu and other radionuclides in ceramic disks, (3) boiling water reactor (BWR) MOX-based spent nuclear fuel, (4) pressurized water reactor (PWR) MOX-based spent nuclear fuel, and (5) immobilized Pu and other radionuclides in glass-bonded zeolite (GBZ). The purpose of this feasibility analysis is to compare the performance of these Pu forms against those currently being considered for disposal in a repository. The comparison for these Pu forms is based on information in the *Report on Evaluation of Plutonium Waste Forms For Repository Disposal*. Further, since the NWPA (as amended) identifies Yucca Mountain, Nevada, as the only location for repository site characterization studies, all candidate waste form performance analyses assume the same geological conditions (unsaturated tuff) as that site.

For each alternative, the total number of additional, if any, waste packages that would be added to the approximately 12,000 packages currently envisioned for the first HLW repository is small enough that any changes in emplacement could be accommodated within the design ratings of such a repository.

H.2 GLASS FORM WITH RADIONUCLIDES

The Pu-loaded glass form is assumed to be fabricated in a new facility using borosilicate glass as the vitrified matrix with the radioactive Cs isotope (Cesium-137) mixed in to provide a source of radiation as a barrier to theft and diversion. This PEIS analyzes only gadolinium (Gd) although boron and lithium neutron absorbers present in the borosilicate glass could be supplemented with samarium or other neutron absorbers.

H.2.1 ASSUMPTIONS

Figure H.2.1–1 shows the waste package containing the glass forms. For the purposes of the PEIS analyses, the following assumptions have been made:

- The waste is packaged as shown in Figure H.2.1–1.
- Molten glass is poured into stainless steel canisters to form encased glass logs that are similar to the Defense Waste Processing Facility (DWPF) glass logs and canisters.
- Each transportation cask holds five of these canisters; each disposal waste package holds four of these canisters.

H.2.2 CHARACTERISTICS

Each proposed glass log from a vitrification facility process consists of 1,540 kilograms (kg) (3,387 pounds [lb]) of borosilicate glass in a stainless steel canister containing 84 kg (185 lb) Pu, 1 kg (2.2 lb) Cs-137, and 55 kg (122 lb) Gd. The Gd, together with the boron and lithium in the glass, acts as a neutron absorber. Other than the addition of Pu, Cs, and Gd, the composition of this glass is assumed to be similar to the borosilicate glass candidate waste form in production at the DWPF at Savannah River Site.

H.2.3 COMPARATIVE ANALYSIS

Regulatory. Any waste form that is accepted for disposal in an HLW geologic repository must comply with the provisions of the NWPA, as amended. According to Section 2(12)A of the NWPA, the definition of HLW does not explicitly include Pu loaded into borosilicate glass. However, under Section 2(12)B of the NWPA, the NRC has the authority to classify this waste as HLW through rulemaking. Such rulemaking or clarification in authorizing legislation will be necessary before this waste form can be considered for disposal in an NWPA repository. The final disposal of this waste form will have to conform to the licensing provisions of 10 CFR 60. Further, it is current policy of the DOE not to accept into the first HLW repository any wastes that include components regulated as hazardous under the *Resource Conservation and Recovery Act* (RCRA) (DOE 1995a:6). The absence of any RCRA-regulated hazardous materials in the final glass form would have to be demonstrated prior to acceptance into the HLW repository.

Criticality. The effective neutron multiplication factor (k_{eff}) for the intact glass form, assuming credit for the neutron absorbers during the post-closure period, is calculated to be to less than 0.3, which is well below the 0.95 maximum value of k_{eff} allowed (10 CFR 60).

Thermal. As shown in Figure H.2.3–1, the results of a thermal analysis of a waste package containing four Pu glass logs indicate that the peak temperature reached by the glass package is about 200 degrees Centigrade (°C) (~400 degrees Fahrenheit [°F]), which is within 5 percent of the peak temperature predicted for the glass logs from the DWPF. These predicted temperatures are far lower than, and therefore safely away from, the glass transition temperature of 400 °C (750 °F). Such small differences in temperature and thermal output are unlikely to materially affect the thermal balance of any repository.

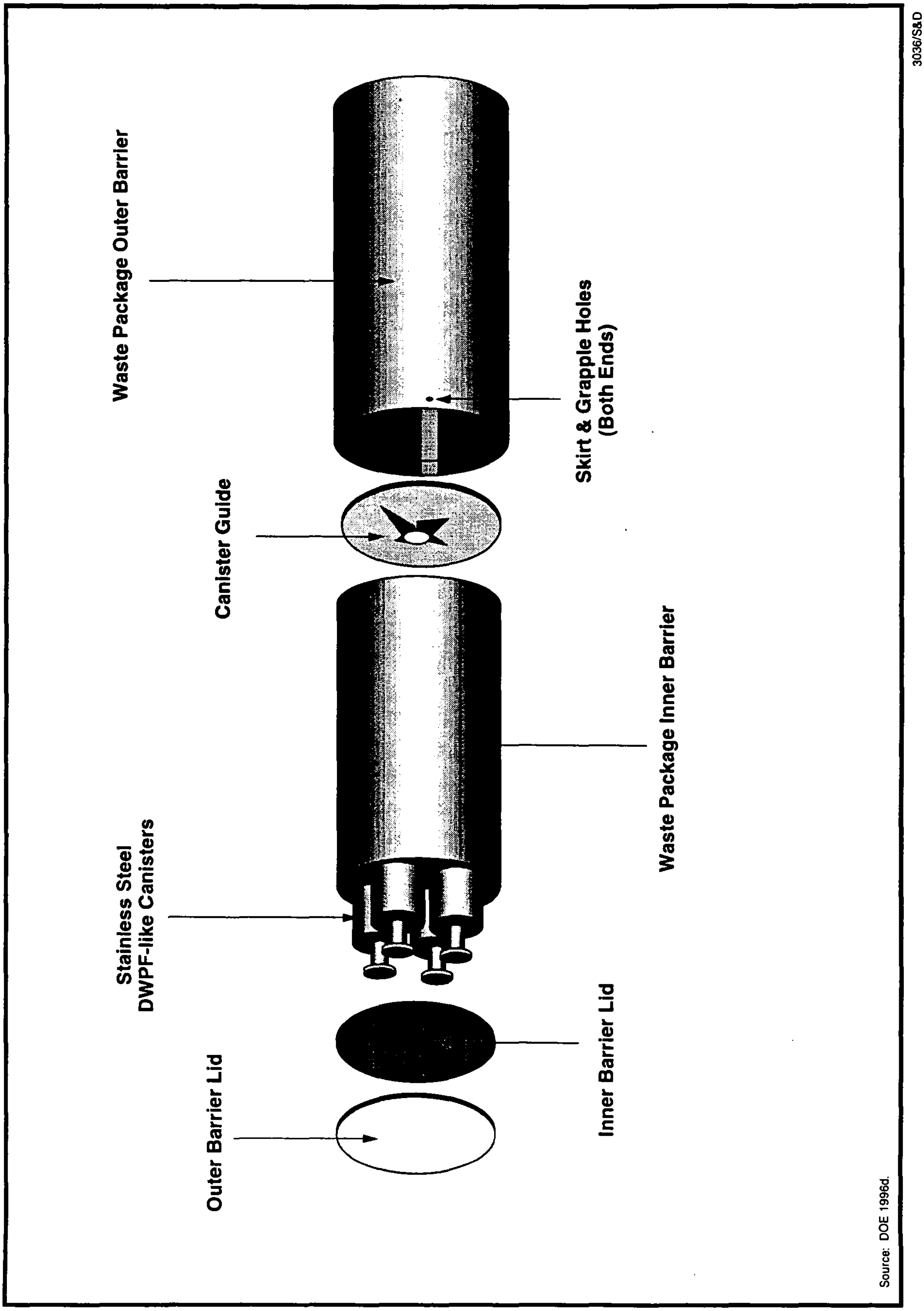

Figure H.2.1–1. Schematic of Waste Package Containing Canisters of Plutonium Immobilized in Glass.

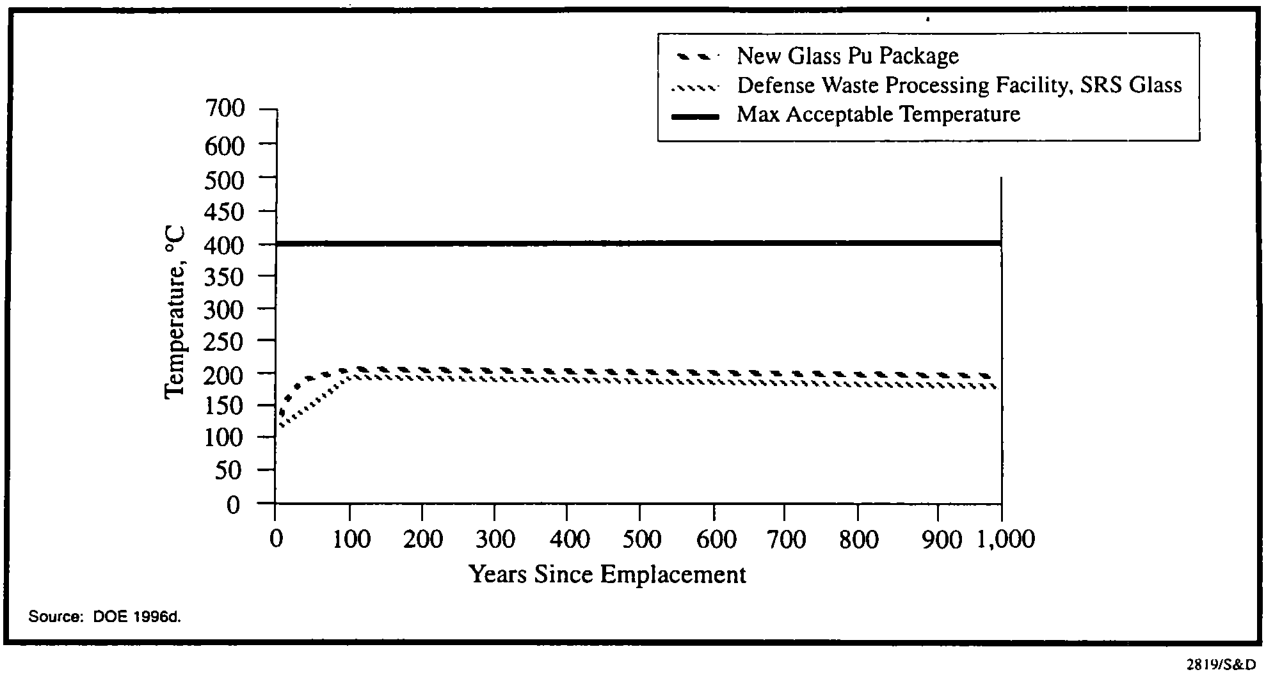

Figure H.2.3–1. Thermal Comparison of Plutonium-Loaded Glass Waste Package Versus Defense Waste Processing Facility Glass Waste Package.

Radiation. A comparison between the DWPF glass and the glass containing Pu shows that the radiation dose at the waste package surface is 81 roentgen equivalent man (rem)/hour (hr) for the DWPF glass compared to 129 rem/hr for the Pu glass. This Pu-glass radiation is above the threshold value for radiolytic corrosion. A 0.4 centimeters (cm) (0.16 inches [in]) additional thickness of the copper-nickel (Cu-Ni) alloy waste package outer barrier would be required to reduce the radiation to an acceptable (100 rad [radiation absorbed dose]/hr) level to protect the waste package from radiolysis-induced corrosion. Additional shielding is also required to protect workers. Doses at a distance of 2 meters (m) (6.6 feet [ft]) from the waste package surface show values of 12.5 rem/hr for the DWPF glass and 25 rem/hr for the Pu glass. For emplacement in the repository, only 5 cm (2 in) of lead thickness and 0.5 cm (0.2 in) of borated polyethylene neutron shielding must be added to the waste package underground transporter to reduce the radiation doses to meet the standard allowable dose of 10 millirem/hr at 2 m (7 ft) from lateral outer surfaces (49 CFR 173.441) to ensure worker protection. An alternative approach to accommodating the higher radiation from the Pu-loaded glass would be to reduce the number of canisters per waste package or the quantity of Cs-137.

Releases. The peak doses from a repository that contains commercial spent nuclear fuel, vitrified HLW, and Pu immobilized in borosilicate glass are the same as from a repository that contains only commercial spent nuclear fuel and vitrified HLW, for periods up to one million years (DOE 1996d:4-12). These results are to be expected since the quantity of Pu glass is small compared to the quantity of spent nuclear fuel in the repository.

H.3 CERAMIC IMMOBILIZED FORMS WITH RADIONUCLIDES

The Pu-loaded ceramic matrix form is assumed to be fabricated in a new facility. As in the vitrification alternative, Cs-137 is mixed in to provide a source of radiation, and Gd acts as a neutron absorber.

H.3.1 ASSUMPTIONS

Figure H.3.1–1 shows the waste package containing the ceramic forms. For the purposes of the PEIS analyses, the following assumptions have been made:

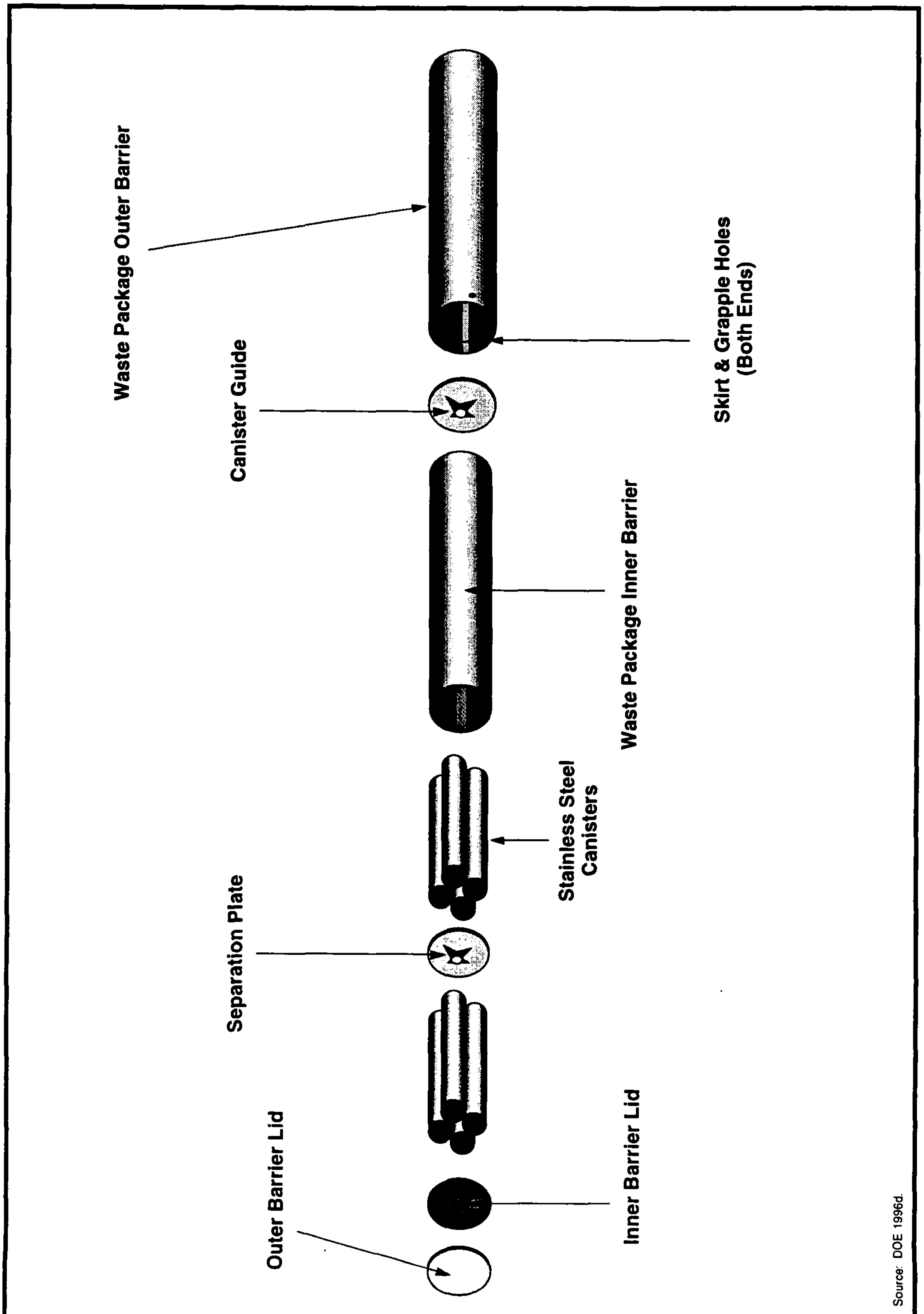

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Figure H.3.1–1. Schematic of Waste Package Containing Canisters of Plutonium Immobilized in Ceramic.

- The waste is packaged as shown in Figure H.3.1–1.

- Ceramic disks will be stacked inside stainless steel canisters. These canisters are similar to the DWPF canisters.

- Each disposal waste package holds eight of these canisters.

H.3.2 CHARACTERISTICS

Each canister of the proposed waste form contains 20 ceramic disks; each disk is approximately 30 cm (12 in) in diameter, and 10 cm (4 in) thick. Each disk has stainless steel plates added to the top and bottom, and a stainless steel shell around the curved surface. The disks are stacked vertically in a stainless steel canister approximately 2.5 m (8 ft) long, and 35 cm (14 in) in diameter. The ceramic disks consist of zirconolite, hollandite, and rutile. For each disk, the zirconolite incorporates 2.6 kg (6.0 lb) of Gd, and the hollandite incorporates 4.0 kg (9.0 lb) of Pu and 0.07 kg (2.0 oz) of Cs-137. The space surrounding the stack of disks inside the canister is filled with titanium oxide powder.

H.3.3 COMPARATIVE ANALYSIS

Regulatory. Any waste form that is accepted for disposal in an HLW geologic repository must comply with the provisions of the NWPA, as amended. According to Section 2(12)A of the NWPA, the definition of HLW does not explicitly include Pu loaded into a ceramic matrix. However, under Section 2(12)B of the NWPA, the NRC has the authority to classify this waste as HLW through rulemaking. Such rulemaking or clarification in authorizing legislation would be necessary before this waste form can be considered for disposal in an NWPA repository. The final disposal of this waste form will have to conform to the licensing provisions of 10 *Code of Federal Regulations* (CFR) 60. Further, it is current policy of the DOE not to accept into the first HLW repository any wastes which include components regulated as hazardous under RCRA (DOE 1995a:6). The absence of any RCRA-regulated hazardous materials in the final ceramic form would have to be demonstrated prior to acceptance into the HLW repository.

Criticality. Preliminary criticality calculations for the intact ceramic waste package, under dry or flooded conditions, and assuming credit for the Gd neutron absorber, yields k_{eff} values of less than 0.7, which is below the 0.95 maximum value of k_{eff} allowed (10 CFR 60).

Thermal. As shown in Figure H.3.3–1, the results of a thermal analysis of Pu-loaded ceramic waste packages shows that peak temperatures are around 200 °C (~400 °F), declining as a function of time. Ceramic, unlike glass, does not have a transition temperature because it is a crystalline material. The lowest melting point temperature for the oxides of this ceramic material is around 1800 °C (3270 °F). Therefore, the calculated peak temperatures are unlikely to affect the ceramic matrix. Further, the temperature differences between the ceramic waste package and the DWPF HLW glass waste package are negligibly small.

Radiation. A comparison between the DWPF HLW glass and the Pu-loaded ceramic shows that the radiation dose at the waste package surface is 81 rem/hr for the DWPF glass compared to 309 rem/hr for the ceramic. The radiation level for the ceramic form is above the threshold value for radiolytic corrosion. Consequently, a 1-cm (0.4-in) additional thickness of the Cu-Ni alloy waste package outer barrier would be required to reduce the radiation to an acceptable level (100 rad/hr) to protect the waste package from radiolytic corrosion. Additional shielding is also required to protect workers. Doses at 2 m (6.6 ft) from the package surface show values of 12.5 rem/hr for the DWPF glass and 56.4 rem/hr for the ceramic. For emplacement in a repository, only 5 cm (2 in) of lead thickness and 0.5 cm (0.2 in) of borated polyethylene neutron shielding must be added to the waste package underground transporter to reduce the radiation doses to meet the standard allowable dose of 10 mrem/hr at 2 m (7 ft) from lateral surfaces (49 CFR 173.441) to ensure worker protection. An alternative approach to

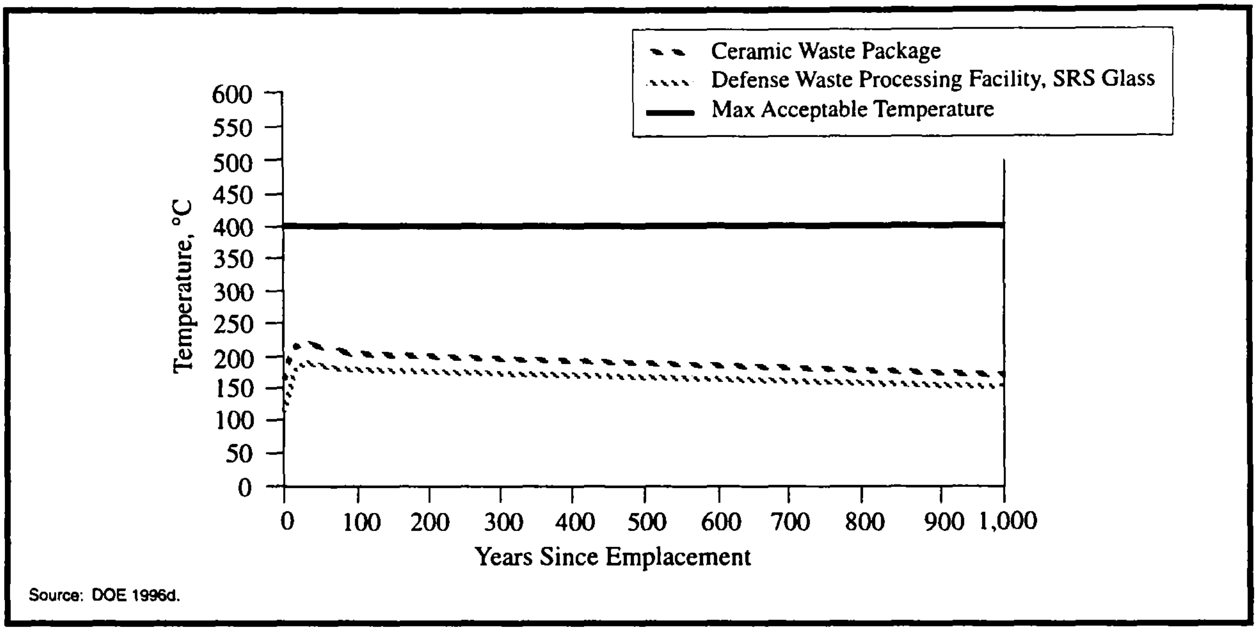

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Figure H.3.3–1. Thermal Comparison of Plutonium-Loaded Ceramic Waste Package Versus Defense Waste Processing Facility Glass Waste Package.

accommodating the higher radiation fields from the Pu-loaded ceramic would be to reduce either the number of canisters per package or the quantity of Cs-137.

Releases. The peak doses from a repository that contains commercial spent nuclear fuel, vitrified HLW, and Pu immobilized in ceramic are the same as from a repository that contains only commercial spent nuclear fuel and vitrified HLW, for periods up to one million years. The difference in dose rates is insignificant between these two cases (DOE 1996d:5-12). These results are to be expected since the quantity of Pu in ceramic is small compared to the quantity of spent nuclear fuel in the repository.

H.4 BOILING WATER REACTOR—MIXED OXIDE BURNING REACTOR SPENT NUCLEAR FUEL FORM

Boiling water reactors are used in existing commercial power generation; therefore, the BWR form of the MOX spent nuclear fuel could be the output product from both the Existing LWR Alternative and the Partially Completed LWR Alternative if the latter is consistent with the BWR design. The performance of this MOX spent fuel is compared to the corresponding commercial BWR uranium-based boiling water reactors spent nuclear fuel.

H.4.1 ASSUMPTIONS

For the purposes of the PEIS analyses, the following assumptions have been made:

- The Pu will be fabricated into MOX nuclear reactor fuel and used for power generation in four boiling water reactors and allowed to cool at the reactor site(s) in the spent fuel pools for at least 10 years before shipment to a repository.

- The spent fuel will be emplaced in large (40 BWR assembly) waste packages for emplacement in a repository.

H.4.2 CHARACTERISTICS

The MOX spent fuel assembly from existing BWRs will have the following characteristics: (1) total Pu of 3.4 kg (7.5 lb), (2) Pu-239 to total Pu ratio of 0.4, (3) total heavy metal content of 172 kg (379 lbs), and (4) burn up of 37.6 gigawatt-days (GWd)/ton (t) of heavy metal. Radiation is analyzed using neutron and gamma source strengths by energy group.

H.4.3 COMPARATIVE ANALYSIS

Regulatory. An HLW repository, if approved under the provisions of the NWPA, would serve primarily as the disposal site for commercial spent nuclear fuel and defense-generated HLW. The MOX spent fuel that would be generated by this alternative falls within the definition of "spent nuclear fuel" per Section 2(23) of the NWPA and could, therefore, be considered a candidate for disposal in an NWPA repository. Licensing for the disposal of this MOX spent fuel form must follow the provisions of 10 CFR 60.

Criticality. Figure H.4.3–1 compares the results of the criticality analyses for a waste package containing all MOX spent fuel with one containing U-based spent fuel. The k_{eff} for the MOX spent fuel is below that of U-based fuels and well below the 0.95 maximum value allowed for k_{eff} (10 CFR 60).

Thermal. Figure H.4.3–2 shows the results of a thermal analysis of the MOX spent fuel element in a fully loaded, emplaced waste package. The peak cladding temperature is below the 350 °C (662 °F) limit required to maintain cladding integrity. Calculations also indicate that for the first 100 years the MOX cladding temperature continues to be lower than that of the corresponding U-based spent fuel. The slightly higher temperatures beyond the 100 years are so small as to have a negligible affect on the thermal balance of any repository.

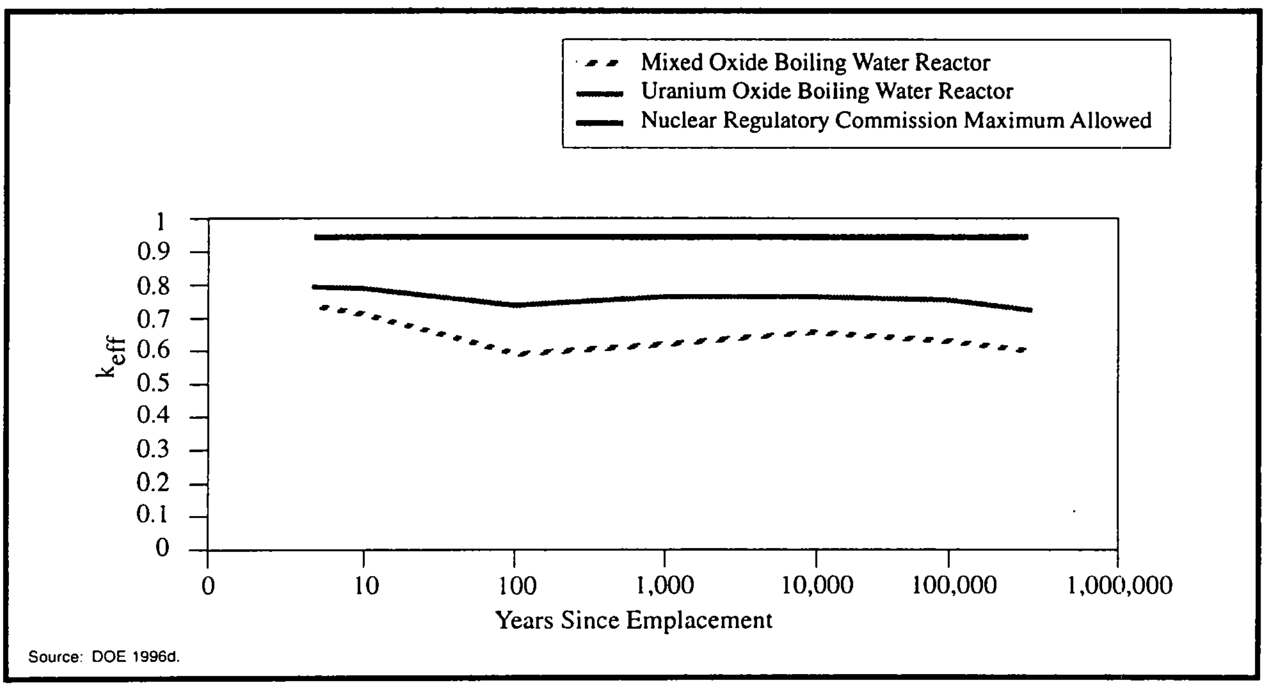

Figure H.4.3–1. Effective Multiplication Factor (k_{eff}) of a Boiling Water Reactor Spent Fuel Waste Package.

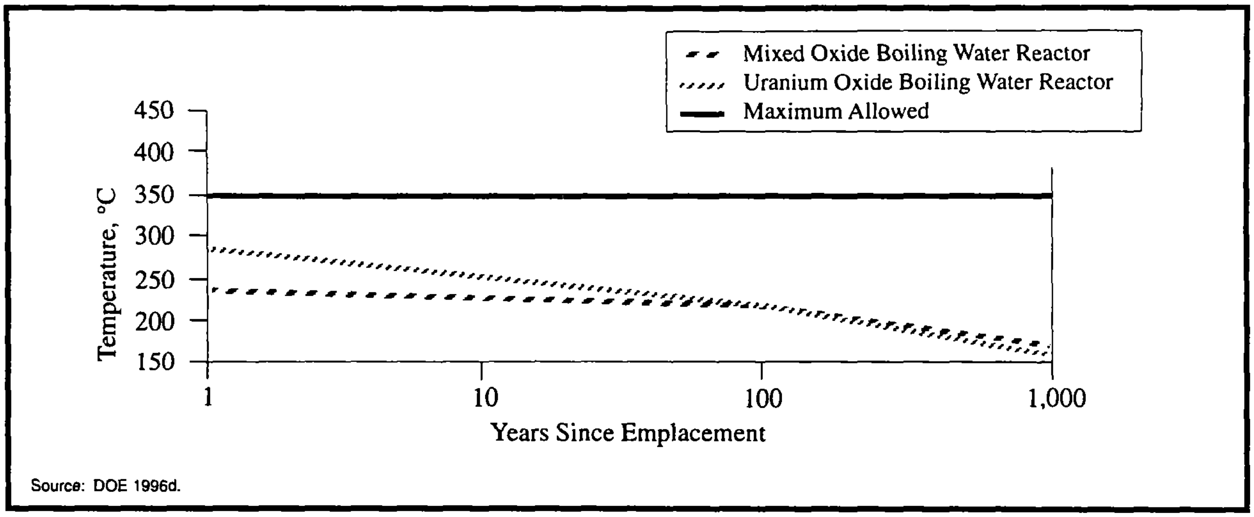

Figure H.4.3–2. ***Thermal Comparison of Peak Cladding Temperature of Boiling Water Reactor Fuel Element Versus Uranium-Based Reactor Fuel Element.***

Radiation. Radiation calculations predict that the unshielded dose rates at a distance of 2 m (6.6 ft) are slightly higher for the waste packages containing MOX fuel than for those containing uranium-based fuels. The gamma radiation dose for the MOX fuel is 5 rem/hr versus 4 rem/hr for the U-based fuel. The neutron radiation dose is 1.54 rem/hr for the MOX as versus to 0.8 rem/hr for the uranium fuel. The higher radiation doses can be accommodated by increasing the transporter shielding thickness by 0.125 cm (0.05 in) of lead for the gamma radiation, and 1.25 cm (0.5 in) of boron-polyethylene for the neutron radiation.

Releases. The calculated doses for just the waste packages of MOX-based spent fuel are 100 times less than that for a repository that contains both MOX and (U-based) commercial spent nuclear fuels (DOE 1996d:3-8). These results support the conclusion that the performance of the repository is dominated by the presence of (U-based) commercial spent nuclear fuel and is expected since the quantity MOX-based spent nuclear fuel is small compared to the larger quantity of commercial spent nuclear fuel in the repository.

H.5 PRESSURIZED WATER REACTOR—MIXED OXIDE BURNING REACTOR SPENT NUCLEAR FUEL FORM

For the Evolutionary LWR Alternative, a PWR could be the design for burning MOX fuel. PWRs are used in existing commercial power plants; therefore, the PWR form of the spent MOX nuclear fuel could be the output product from the Evolutionary LWR Alternative, Existing LWR Alternative, and the Partially Completed LWR Alternative if the latter reactors are consistent with the PWR design. The performance of this MOX spent nuclear fuel is compared to the corresponding U-based PWR spent nuclear fuel.

H.5.1 ASSUMPTIONS

For the purposes of the PEIS analyses, the following assumptions have been made:

- The Pu will be fabricated into MOX nuclear fuel and used for power generation in two PWRs and allowed to cool at the reactor site(s) in the spent fuel pools for at least 10 years before shipment to a repository.

- The spent fuel will be emplaced in large waste packages for emplacement in a repository.

H.5.2 CHARACTERISTICS

The MOX spent fuel assembly from an evolutionary PWR will have the following characteristics: (1) total Pu of 20 kg (44 lb), (2) Pu-239 to total Pu ratio of 0.6, (3) total heavy metal content of 410 kg (900 lbs), and (4) burn up of 43 GWd/t of heavy metal. Radiation is analyzed using neutron and gamma source strengths by energy group.

H.5.3 COMPARATIVE ANALYSIS

Regulatory. An HLW repository, if approved under the provisions of the NWPA, would serve primarily as the disposal site for commercial and DOE-owned spent nuclear fuel and HLW. The MOX spent fuel that would be generated by this alternative falls within the definition of "spent nuclear fuel" per Section 2(23) of the NWPA and could, therefore, be considered a candidate for disposal in an NWPA repository. Licensing for the disposal of this MOX spent fuel form must follow the provisions of 10 CFR 60.

Criticality. Calculations for a MOX PWR spent fuel waste package show that to maintain a value below the 0.95 maximum value allowed for k_{eff} (10 CFR 60), the waste package can hold only four assemblies. This calculation assumed no additional criticality control technology. Should such technology be applied (for example, disposable control rod assemblies were added to the waste packages) calculations show that 21 assemblies could be loaded in each waste package. For either the 4 or 21 assemblies/waste package case, the k_{eff} value is expected to decline with time in a manner similar to that of the BWR spent fuel waste package as shown in Figure H.4.3–1.

Thermal. Figure H.5.3–1 shows the results of a thermal analysis of the MOX PWR spent fuel. The peak cladding temperature is below the 350 °C (662 °F) limit required to maintain cladding integrity. For the first 100 years the temperature also remains lower than that of the corresponding U-based spent fuel. The additional heat from all the spent nuclear fuel packages produced by the PWRs would be so small as to have a negligible affect on the thermal balance of any repository.

Radiation. Radiation calculations for the 21-assembly MOX PWR waste package shows that the higher dose rates from the MOX package (compared to the package containing U-based spent fuel) can be easily accommodated by increasing the transporter shielding thickness by 0.4 cm (0.16 in) of lead for the gamma radiation, and 1.25 cm (0.5 in) of boron-polyethylene for the neutron radiation. The shielding thickness requirements for a four-assembly package will be less than these values.

Releases. The calculated doses for just the waste packages of MOX-based spent fuel are 100 times less than that for a repository that contains both MOX and (U-based) commercial spent nuclear fuel (DOE 1996d:6-7). These results support the conclusion that the performance of the repository is dominated by the presence of (U-based) commercial spent nuclear fuel and is expected since the quantity of MOX-based spent fuel is small compared to the larger quantity of commercial spent fuel in the repository.

H.6 GLASS-BONDED ZEOLITE FORM

The Pu-loaded GBZ form is assumed to be fabricated in an electrometallurgical treatment process and has characteristics for long term disposability in a repository that are similar to the borosilicate glass produced in the DWPF (DOE 1996d:4-1). The GBZ waste form constitutes another immobilization alternative which would require disposal in a repository.

H.6.1 ASSUMPTIONS

For the purposes of the PEIS analyses, the following assumptions have been made:

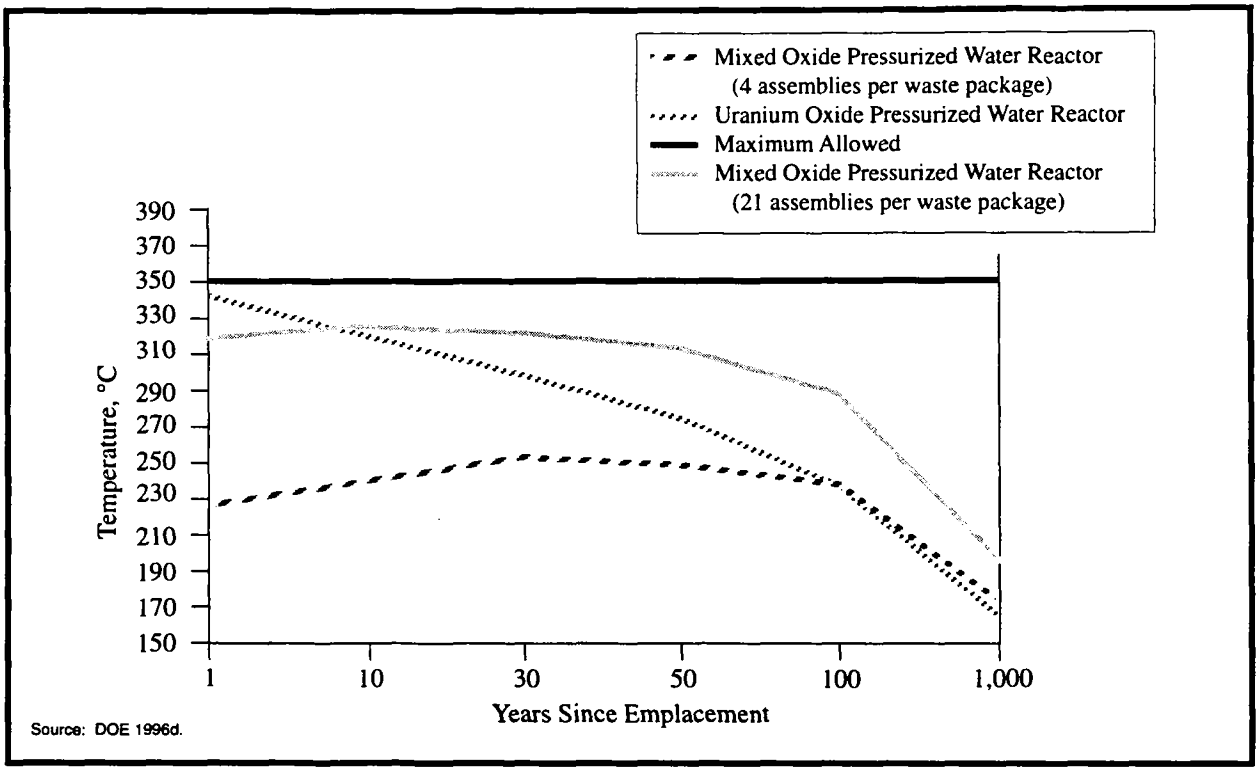

Figure H.5.3–1. Thermal Comparison of Peak Cladding Temperature of Pressurized Water Reactor Fuel Element Versus Uranium-Based Reactor Fuel Element.

- The waste form is packaged in DWPF-like canisters.
- Each transportation cask holds five of these canisters; each disposal waste package holds four of these canisters.

H.6.2 CHARACTERISTICS

The GBZ will be prepared by sorbing a molten chloride Pu salt on an anhydrous zeolite, which is then blended with a glass frit. The whole mixture is heated in a mold to above the glass transition temperature and pressed to bond the zeolite to the glass. The chemical constituents of the waste form are as follows: 52 kg (114 lbs) of Pu, 21 kg (46 lbs) of Gd, 5 kg (11 lbs) of Cs, 364 kg (800 lbs) of zeolite, and 520 kg (1,144 lbs) of borosilicate glass. The remainder is made up of barium, lithium, potassium, sodium, and chlorides.

H.6.3 COMPARATIVE ANALYSIS

Regulatory. Any waste form that is accepted for disposal in an HLW geologic repository must comply with the provisions of the NWPA, as amended. According to Section 2(12)A of the NWPA, the definition of HLW does not explicitly include Pu loaded into GBZ. However, under Section 2(12)B of the NWPA, the NRC has the authority to classify this waste as HLW through rulemaking. Such rulemaking or clarification in authorizing legislation will be necessary before this waste form can be considered for disposal in an NWPA repository. The final disposal of this waste form will have to conform to the licensing provisions of 10 CFR 60. Further, it is current policy of the DOE not to accept into the first HLW repository any wastes that include components regulated as hazardous under RCRA (DOE 1995a:6). The absence of any RCRA-regulated hazardous materials in the final GBZ form would have to be demonstrated prior to acceptance into the HLW repository.

Criticality. Preliminary criticality calculations show the Pu-loaded GBZ in a dry, intact configuration has a k_{eff} of less than 0.2, which is less than the borosilicate glass form primarily because of the lower total Pu content of each waste package containing the GBZ. The k_{eff} values for the GBZ under flooded conditions are less than half the 0.3 value (0.13) calculated for the borosilicate glass form and well below the 0.95 maximum value allowed for k_{eff} (10 CFR 60).

Thermal. Because the Pu concentration in a waste package containing Pu-loaded GBZ canisters is 80 percent of the Pu concentration in a package containing Pu-loaded borosilicate glass, and because the peak temperature reached by borosilicate glass is below the 400 °C (750 °F) glass transition value, the peak temperature for the GBZ is also expected to be below the 400 °C (750 °F) glass transition temperature. More specifically, the lower Pu content of the GBZ means that the heat generation at 40 years and beyond would be much smaller than for the Pu-loaded borosilicate glass.

Radiation. A comparison between the DWPF HLW glass and the Pu-loaded GBZ shows that the radiation dose at the waste package surface is 81 rem/hr for the DWPF glass compared to 120 rem/hr for the Pu-loaded GBZ. Since the radiation level for GBZ is above the threshold value of 100 rads/hr for radiolytic corrosion, the waste package outer barrier thickness would need to be increased by 0.3 cm (0.11 in). Additional shielding is also required to protect workers. The dose rate at 2 m (6.6 ft) from the waste package is 23 rem/hr for the GBZ versus 12.5 rem/hr for DWPF glass. As in the case for the Pu-loaded borosilicate glass form, the addition of 5 cm (2 in) of lead shielding to the underground transporter would reduce the radiation doses to meet the standard allowable dose of 10 mrem/hr at 2 m (6.6 ft) from lateral outer surfaces (49 CFR 173.441) to ensure worker protection.

Releases. The peak doses from a repository that contains commercial spent nuclear fuel, vitrified HLW, and Pu immobilized in GBZ are the same as from a repository that contains only commercial spent nuclear fuel and vitrified HLW, for periods up to 1 million years (DOE 1996d:7-9). These results are to be expected since the quantity of Pu in GBZ is small compared to the quantity of spent nuclear fuel in the repository.